[7590-01-P]

NUCLEAR REGULATORY COMMISSION

10 CFR Parts 50 and 52

[Docket No. PRM-50-105; NRC-2012-0056]

In-core Thermocouples at Different Elevations and Radial Positions in Reactor Core

AGENCY: Nuclear Regulatory Commission.

ACTION: Petition for rulemaking; denial.

SUMMARY: The U.S. Nuclear Regulatory Commission (NRC) is denying a petition for rulemaking (PRM), PRM-50-105, submitted by Mark Leyse (the petitioner) on February 28, 2012. The petitioner requested that the NRC require all holders of operating licenses for nuclear power plants (NPPs) to operate NPPs with in-core thermocouples at different elevations and radial positions throughout the reactor core to enable the operators to accurately measure a large range of in-core temperatures in NPP steady-state and transient conditions. The NRC is denying the PRM because: there are no protection or plant control functions that utilize inputs from core exit thermocouples (CETs); there is no operational necessity for more accurate measurement of temperatures throughout the core; the petition provided inadequate justification of why precise knowledge of core temperature at various elevations and radial positions would enhance safety or change operator action; and the NRC

believes that, despite the known limitations of CETs, CETs are sufficient to allow NPP operators to take timely and effective action in the event of an accident.

DATES: The docket for the petition for rulemaking, PRM-50-105, is closed on **[INSERT DATE OF PUBLICATION]**.

ADDRESSES: Please refer to Docket ID NRC-2012-0056 when contacting the NRC about the availability of information for this petition. You may access information related to this petition by any of the following methods:

- **Federal Rulemaking Web Site:** Go to http://www.regulations.gov and search on Docket ID NRC-2012-0056. Address questions about NRC dockets to Carol Gallagher, telephone: 301-492-3668; e-mail: Carol.Gallagher@nrc.gov.

- **The NRC's Agencywide Documents Access and Management System (ADAMS):** You may access publicly available documents online in the NRC Library at http://www.nrc.gov/reading-rm/adams.html. To begin the search, select "ADAMS Public Documents" and then select "Begin Web-based ADAMS Search." For problems with ADAMS, please contact the NRC's Public Document Room (PDR) reference staff at 1-800-397-4209, 301-415-4737, or by e-mail to PDR.Resource@nrc.gov. The ADAMS Accession Number for each document referenced in this document (if that document is available in ADAMS) is provided the first time that a document is referenced. In addition, for the convenience of the reader, the ADAMS Accession Numbers are provided in a table in Section V, "Availability of Documents," of this document.

- **The NRC's PDR:** You may examine and purchase copies of public documents at the NRC's PDR, O1-F21, One White Flint North, 11555 Rockville Pike, Rockville, Maryland 20852.

FOR FURTHER INFORMATION CONTACT: Tara Inverso, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, telephone: 301-415-1024; e-mail: Tara.Inverso@nrc.gov.

SUPPLEMENTARY INFORMATION:

I. Background.

II. NRC Technical Evaluation.

III. Public Comments on the Petition.

IV. Ongoing NRC Activities Related to Reactor and Containment Instrumentation.

V. Availability of Documents.

VI. Determination of the Petition.

I. Background.

The NRC received a petition for rulemaking (ADAMS Accession No. ML12065A215) on February 28, 2012, and assigned it Docket No. PRM-50-105. The NRC published a notice of receipt and request for public comment in the *Federal Register* (FR) on May 23, 2012 (77 FR 30435).

The petitioner requested that the NRC amend its regulations in Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR), "Domestic Licensing of Production and Utilization Facilities," to require all holders of operating licenses for NPPs to operate NPPs with in-core thermocouples at different elevations and radial positions throughout the reactor core to enable NPP operators to accurately measure a large range of in-core temperatures in NPP steady-state and transient conditions. The petitioner asserted that, in the event of a severe accident, in-core thermocouples would provide NPP operators with crucial information to help operators manage the accident. In support of the petition, the petitioner cited several reports and findings, including the Report of the President's Commission on the Accident at Three Mile Island [TMI]: “The Need for Change: The Legacy of TMI,” dated October 1979. The petitioner asserted that “[i]n the last three decades, NRC has not made a regulation requiring that NPPs operate with in-core thermocouples at different elevations and radial positions throughout the reactor core to enable NPP operators to accurately measure a large range of in-core temperatures in NPP steady-state and transient conditions, which would help fulfill the President's Commission recommendations.” The petitioner further stated that, if another severe accident were to occur in the United States, NPP operators would not know what the in-core temperatures would be during the progression of the accident, and concluded that, in a severe accident, core-exit thermocouples would be the primary tool used to detect inadequate core cooling and core uncovery.

II. NRC Technical Evaluation.

The petitioner requested that the NRC require in-core thermocouples be installed in all NPPs; this would include both pressurized water reactors (PWRs) and boiling water reactors (BWRs). However, BWRs do not use CETs, and thermocouple response in BWR applications is not currently known. Furthermore, the experiments referenced throughout the PRM studied only PWRs. Because the issues and arguments raised in the PRM do not apply to BWRs, and because the PRM does not list any limitations on BWR instrumentation, there is no basis provided to evaluate this PRM for BWRs. Therefore, the NRC is evaluating this PRM as it pertains to PWRs only.

During normal operation in a PWR, reactor coolant system (RCS) hot leg and cold leg temperatures are the primary indications of core condition. Measurements of RCS hot and cold leg temperatures from safety-related instrumentation provide the necessary input to a plant’s reactor protection system. There are no reactor protection or plant control functions that use inputs from the CETs. Additionally, the CETs are not the only source of information relied on to initiate reactor operator responses to accident conditions. The uses of CETs will be described in more detail, as part of the NRC's evaluation of the issues raised in the PRM with respect to the use of CETs.

PRM Issue 1: Core Exit Thermocouple Limitations

The petitioner stated that, "in a severe accident, in many cases, a predetermined core exit temperature measurement (e.g., 1200 °F) would be used to signal the time for NPP operators to transition from EOPs [Emergency Operating Procedures] to implementing SAMGs [Severe Accident Management Guidelines]." However, experimental data indicates that CET measurements have significant limitations. A report[1] prepared by the Organization for Economic Cooperation and Development (OECD) Nuclear Energy Agency (NEA), Committee on the Safety of Nuclear Installations, entitled, "Core Exit Temperature (CET)[2] Effectiveness in Accident Management of Nuclear Power Reactor," dated November 26, 2010, concluded:

- The use of CET measurements has limitations in detecting inadequate core cooling and core uncovery,
- The CET indication displays in all cases a significant delay (up to several hundred [seconds]), and
- The CET reading is always significantly lower (up to several 100 [Kelvin]) than the actual maximum cladding temperature.

The petition asserted that the NRC and the nuclear industry have ignored experimental data indicating that CET measurements have significant limitations. The results of four tests

[1] *Available at* http://www.oecd-nea.org/nsd/docs/2010/csni-r2010-9.pdf.
[2] Note that the OECD report uses the acronym CET to refer to core exit temperature, but the NRC uses the acronym CET to refer to core exit thermocouples in this document.

performed in the loss-of-fluid test (LOFT) facility show that: 1) there was a delay between the core uncovery and the thermocouple response, and 2) the measured core exit thermocouple response was several hundred Kelvin lower than the maximum cladding temperatures in the core. The petitioner cited NUREG/CR-3386, "Detection of Inadequate Core Cooling with Core Exit Thermocouples: LOFT PWR Experience," dated November 1983 (ADAMS Accession No. ML13032A566), which states: "There may be accident scenarios in which these [thermocouples] would not detect inadequate core cooling that preceded core damage."

The NRC reviewed PRM Issue 1 and acknowledges that the CET limitations cited by the petitioner are extensively documented in test reports from the identified experimental programs. However, while these test programs were conducted at large-scale test facilities appropriately scaled (using a power to volume relationship) to produce thermal-hydraulic phenomena similar to phenomena that could occur in a commercial PWR, the scaling distortions introduced by the facilities and the effects of plant-specific CET installation methods preclude the direct extrapolation of the test results to reactor scale. In fact, the same OECD report referenced by the petitioner also states:

> Qualitative application/extrapolation of the CET response to reactor scale is possible. However, direct extrapolation in quantitative terms to the reactor scale should be avoided in general or done with special care due to limitations of the experimental facilities in terms of geometrical details, unavoidable distortion in the scaling of the overall geometry, and of the heat capacity of structures.

The NRC views these results within the context of their applicability to full-scale plants in order to use the data to assess the capability of the computer models used to perform full-plant simulations. The separate test facilities, such as LOFT and Primarkreislauf Test Facility Project (PKL), are simulated using computer models, and the results from the simulations are compared with the corresponding data. Once sufficient agreement between the simulation and the data is achieved, or consistent biases are determined, a full-plant simulation can be performed and more definitive, quantitative statements about CET performance can be made. Therefore, these experimental results cannot be, and are not intended to be, quantitatively extrapolated to full-scale plants, as suggested in the petition.

During normal operation, RCS hot leg and cold leg temperatures are the primary indications of core condition. Measurements of RCS hot and cold leg temperatures from safety-related instrumentation provide the necessary input to a plant's reactor protection system. There are no reactor protection or plant control functions that use inputs from the CETs.

During accident conditions, the most significant functions provided by CETs are the determination of a trend in RCS sub-cooling and the known correlation of the indicated temperature to general core conditions for the purposes of identifying the onset of core damage (i.e., a severe accident). For these purposes, the CETs provide the indication necessary to make operational decisions with respect to core damage and perform these essential functions within the expected useful range. In the initial stages of an accident, CETs provide accurate indication of core temperatures for the purposes of determining sub-cooling margin when forced circulation has been lost and confirming that the core remains covered. As an event progresses, CETs provide an indication of initial stages of core damage and are generally used as an entry condition and diagnostic tool during implementation of SAMGs.

Upon entry into the SAMGs, core exit temperature is used as one indication in a diagnostic process to determine core damage; other indications include: RCS level, RCS pressure, containment pressure, containment hydrogen concentration, nuclear instrumentation, and containment high range radiation monitors. As CET readings rise above 1200 °F, it becomes likely that the temperature for some sections of cladding will have exceeded 1800 °F, and therefore it can be assumed that core damage has commenced. With this determination, actions to restore key safety functions will continue in order to restore core cooling and to ensure that fission product barriers remain intact. At no point, either during diagnosis or follow-on actions to restore core cooling, is there an operational necessity for an exact measurement of core temperatures at various locations throughout the core. The petitioner did not provide explicit examples where knowing more precise temperatures would result in more effective operator action. Further, the NRC's evaluation of this petition and relevant information did not reveal added insights on how knowing precise in-core temperatures would result in more effective operator action in a core damage sequence. The correlation between CET readings and fuel cladding temperature, in conjunction with other indications, is sufficient for determining the onset of fuel damage and the need for operator action. Actions taken to restore core cooling would not depend upon a precise measurement of in-core temperature. As the accident progresses, core vessel breach determination is primarily made by utilizing containment pressure and containment radiation indications, and nuclear instrumentation. Core exit thermocouple indications are not used for this determination.

After considering the functions and indications provided by CETs in normal and accident conditions, the NRC determined that the CETs provide adequate indications for their intended purpose.

PRM Issue 2: Nuclear Power Plant Operators' Use of In-Core Thermocouples

The petition asserted that, in the event of a severe accident, in-core thermocouples would enable NPP operators to accurately measure in-core temperatures better than CETs, and would provide crucial information to help operators manage the accident; one example is an indication that it is time to transition from EOPs to implementing SAMGs. Therefore, the petition requested that all holders of operating licenses for NPPs operate NPPs with in-core thermocouples at different elevations and radial positions throughout the reactor core to enable NPP operators to accurately measure a large range of in-core temperatures in NPP steady-state and transient conditions.

As previously stated BWRs do not use CETs, and thermocouple response in BWR applications is not currently known. Furthermore, the experiments referenced throughout the PRM studied only PWRs. Therefore, the NRC is evaluating this PRM as it pertains to PWRs only. The NRC further notes that, in BWRs, saturation conditions exist within the reactor vessel and fuel temperatures are closely related to the saturation pressure. Under accident conditions, reactor vessel water level is the best indication of conditions relating to imminent core damage and drywell radiation monitors are typically the primary method for determining core damage and SAMG entry conditions. For BWRs, SAMG entry conditions are also tied to parameters such as water level, containment hydrogen concentration, and component failures. With regard to PWRs, CETs are located at various radial positions. Therefore, the intent of the petitioner's request to account for various radial temperatures is addressed by the current design.

The petition does not specify any benefit the data from in-core thermocouples could provide or how that benefit would be greater than that provided by core exit thermocouples. As discussed earlier, the limitations of CETs are already well understood and accounted for in existing SAMGs. The benefit provided by CETs, even in recognition of their limitations, is discussed in greater detail in the NRC response to PRM Issue 1. Furthermore, the petitioner cited no actions that would be driven by the additional information obtained from in-core thermocouples.

It is also important to note that the same OECD document referenced by the petitioner contains additional information that provides a perspective that is different from that of the petitioner. For example, from page 48 of the report:

> The conduct of the experiment was rather complicated with repeated openings of two blowdown lines. The timeline for the experiment was thus not very representative of a real accident. . . .Measured cladding temperatures exceeded 2100 K . . .The temperatures were in excess of 2100K for several minutes and the peak temperatures were probably several hundred degrees higher than that. Material examinations showed material formations consistent with temperatures in the range of 2800 K and in local areas over 3000 K.

“An Account of the OECD LOFT Project” of this experiment (LP-FP-2)[3] additionally states on page 53:

> Thermocouples used in the CFM [Center Fuel Module] were calibrated as high as 2100 K. However, many of the CFM temperature measurements were affected by thermocouple cable shunting effects [formation of a new thermocouple junction due to exposure to high temperature] before the temperature at the thermocouple location reached 2100 K.

These statements indicate that in-core thermocouples may not be any more accurate than, or as reliable as, the core exit thermocouples currently used in PWRs, and that they may be subject to additional limitations. It is impractical to mount thermocouples to the fuel cladding surface or fuel spacers. Reactor vessel head modifications would be necessary, as well as the addition of a significant amount of instrumentation wiring and support structures. Furthermore, the addition of in-core thermocouples and the associated supporting components would likely result in significant adverse effects on fluid flow in the core. For instance, fin effects would disturb temperature profiles within the core, and could create calibration difficulties. In addition, installing in-core thermocouples could increase loose parts potential, independence and separation issues, and seismic considerations.

While the previous discussion applies to fuel-cladding-surface-mounted thermocouples, the NRC also considered the petitioner’s request as it may relate to a requirement to install thermocouples in bulk coolant areas within the fuel matrix, such as within instrument tubes. Extensive research has been performed to characterize the relationship between liquid and vapor temperatures and heat transfer rates in the dispersed flow regime expected within the core during severe accident conditions. Significant temperature differences can exist between the bulk coolant, which would contain droplets of liquid water at saturation conditions, and the fuel cladding surface. R. S. Dougall and W. M. Rohsenow, for instance, characterized surface

[3] *Available at* http://www.oecd-nea.org/nsd/reports/OECD_LOFT_final_report_T3907_May1990.pdf.

temperatures that exceeded saturation temperatures by 400 to 700 degrees Fahrenheit in their experimental work.[4] Subsequent work has validated Dougall's and Rohsenow's findings. Because of the significant temperature differences that can exist within the post-accident core region, thermocouples located within the instrument tubes would provide information that offers no greater benefit than that provided by the CETs.

For these reasons, the NRC determined that, for operating PWRs, in-core thermocouples are not necessary, nor would they help operators manage an accident. In addition to these reasons, the NRC notes that the installation and maintenance associated with in-core thermocouples would result in higher doses to plant workers, with no added safety benefit.

The petition requested that the requirement for in-core thermocouples be applied to "all holders of operating licenses for [nuclear power plants]." The NRC interprets this request as applying to both current and future holders of operating licenses under 10 CFR Part 50, as well as current and future holders of combined licenses under 10 CFR Part 52. The NRC believes that this is a reasonable interpretation, inasmuch as combined licenses under 10 CFR Part 52 combine the authority provided under a construction permit and an operating license (albeit with certain conditions and restrictions as set forth in 10 CFR Part 52, Subpart C[5]) into one license. In addition, because the two existing combined licenses reference the AP1000 design certification rule (10 CFR Part 52, Appendix D), which controls the design of the reactor instrumentation, including the placement of thermocouples, the NRC interprets the petition as a request to amend the AP1000 design certification rule.

---

[4] R. S. Dougall and W. M. Rohsenow, "Film Boiling on the Inside of Vertical Tubes with Upward Flow of the Fluid at Low Qualities," 1963, *available at* http://hdl.handle.net/1721.1/62142.

[5] The conditions and limitations of a combined license issued under 10 CFR Part 52 are consistent with, and are intended to comply with, the statutory requirements for combined licenses in Section 185b of the Atomic Energy Act of 1954, as amended.

Because the core of the AP1000 design is similar to the PWRs described throughout this document, the NRC's evaluation of, and determination on, this PRM with respect to PWRs also applies to the AP1000 design and no changes to the AP1000 design are necessary.

PRM Issue 3: Post-Three Mile Island Accident Actions

The petition included a citation from an October 1979 recommendation from the President's Commission on the Three Mile Island Accident, which stated:

> Equipment should be reviewed from the point of view of providing information to operators to help them prevent accidents and to cope with accidents when they occur. Included might be instruments that can provide proper warning and diagnostic information; for example, the measurement of the full range of temperatures within the reactor vessel under normal and abnormal conditions.

The petitioner asserted that the NRC has not made a regulation requiring NPPs to operate with in-core thermocouples at different elevations and radial positions throughout the reactor core to enable NPP operators to accurately measure a large range of in-core temperatures in NPP steady-state and transient conditions, which the petitioner avows would help fulfill the President's Commission's recommendations. The petitioner further asserted that if another severe accident were to occur in the United States, NPP operators would not know what the in-core temperatures were during the progression of the accident.

Following the accident at TMI, the NRC ordered a broad range of safety enhancements at U.S. NPPs. These enhancements include sub-cooled margin monitors, post-accident monitoring instrumentation systems (including CET indications available to operators), and the reactor vessel level monitoring system. These enhancements, combined with other post-TMI requirements for enhanced EOPs and operator training, form part of the Agency's response to the recommendation of the President's Commission on the Three Mile Island Accident.

Regarding the President's Commission's example of "measurement of the full range of temperatures within the reactor vessel under normal and abnormal conditions," evidence of the

NRC's consideration of in-core thermocouples may be found in NUREG-0737, "Clarification of TMI Action Plan Requirements" (ADAMS Accession No. ML051400209), Section II.F.2, "Instrumentation for Detection of Inadequate Core Cooling (ICC)." Item (6) on page 3-114 under Clarifications states:

> The indication must cover the full range from normal operation to complete core uncovery. For example, water-level instrumentation may be chosen to provide advanced warning of two-phase level drop to the top of the core and could be supplemented by other indicators such as incore and core-exit thermocouples provided that the indicated temperatures can be correlated to provide indication of the existence of ICC [inadequate core cooling] and to infer the extent of core uncovery. Alternatively, full-range level instrumentation to the bottom of the core may be employed in conjunction with other diverse indicators such as core-exit thermocouples to preclude misinterpretation due to any inherent deficiencies or inaccuracies in the measurement system selected.

The alternative noted in this excerpt, to use full-range level indication combined with core exit thermocouples, was ultimately the preferred option. Part of the consideration to use the alternative may be found in the NRC's stated position on ICC that requires unambiguous, easy-to-interpret indication of ICC. The NRC chose to use process variables that map directly to clear, easy-to-interpret emergency operating procedures to elicit safe and consistent operator responses to accident scenarios.

<u>PRM Issue 4: Consideration of Experimental Data</u>

The petitioner asserted that the NRC and Westinghouse do not consider that experimental data at four facilities (LOFT, PKL, Rig of Safety Assessment Large-Scale Test Facility (ROSA/LSTF), and OECD/NEA computer codes validation project (PSB-VVER)) indicate that CET measurements would not be an adequate indicator for when to transition from EOPs to implementing SAMGs in a severe accident. The petition listed 13 conclusions from the OECD report that are common to the evaluation of the tests in all four facilities summarized by that report:

- “The use of CET measurements has limitations in detecting inadequate core cooling and core uncovery;”

- “The CET indication displays in all cases a significant delay (up to several 100 [seconds]);”

- “The CET reading is always significantly lower (up to several 100 [Kelvin]) than the actual maximum cladding temperature;”

- “CET performance strongly depends on the accident scenarios and the flow conditions in the core;”

- “The CET reading depends on water fall-back from the upper plenum (due to: e.g., reflux condensing [steam generator] mode or water injection) and radial core power profiles. During significant water fall-back the heat-up of the CET sensor could even be prevented;”

- “The colder upper part of the core and the cold structures above the core are contributing to the temperature difference between the maximum temperature in the core and the CET reading;”

- “The steam velocity through the bundle is a significant parameter affecting CET performance;”

- “Low steam velocities during core boil-off are typical for [small-break loss-of-coolant accident] transients and can advance 3D flow effects;”

- “In the core as well as above (i.e., at the CET measurement level) a radial temperature profile is always measured (e.g., due to radial core power distribution and additional effects of core barrel and heat losses);”

- “Also at low pressure (i.e., shut down conditions) pronounced delays and temperature differences are measured, which become more important with faster core uncovery and colder upper structures;”

- "Despite the delay and the temperature difference the CET reading in the center reflects the cooling conditions in the core;"
- "Any kind of [accident management] procedures using the CET indication should consider the time delay and the temperature difference of the CET behavior;" and
- "In due time after adequate core cooling is re-established in the core the CET corresponds to no more than the saturation temperature."

Finally, the petitioner continued to reference the OECD report, stating that, during the LOFT LP-FP-2 experiment when maximum core temperatures were measured to exceed 3300 °F, CETs were typically measured at 800 °F (more than 2500 °F lower than the maximum core temperatures). He provided that "during the rapid oxidation phase the CET appeared essentially to be disconnected from core temperatures."

The NRC and the industry have long acknowledged the limitations of CETs, but conclude that the use of CETs remains appropriate and would help operators to manage an accident. This awareness is documented in several reports, such as "Limitations of Detecting Inadequate Core Cooling" (U.S. Department of Energy's Office of Scientific and Technical Information ID 6797561) published in 1984 and WCAP-14696-A, Revision 1, "Westinghouse Owners Group Core Damage Assessment Guidance," dated July 1996 (ADAMS Accession No. ML993490267). The delayed indication would not necessarily be a concern during a severe accident. First, the NPP staff relies on other indications to diagnose conditions, such as the reactor vessel level instrumentation system, hot-leg resistance temperature detectors, and containment hydrogen and radiation monitors. Second, whereas the CET indication delay may be up to a few minutes, post-accident operator actions are determined and implemented on a scale that exceeds several minutes. On this time scale, the noted time delay is acceptable.

The petition cited a number of conclusions about CET deficiencies that were noted in the OECD report, and cited on page 8 of the PRM, but the PRM did not specifically acknowledge the following statement from page 129 of the OECD report: "Despite the delay and the temperature difference the CET reading in the center reflects the cooling conditions in the core." It is the NRC's position that scaling challenges, described earlier in this document, exist when extrapolating the results to a full-scale NPP, and these challenges tend to exacerbate the extent of the CET deficiencies cited in the experimental results. Therefore, while the noted deficiencies should be considered qualitatively, overall, in terms of plant applicability, the CETs performed the intended function, as described in the NRC's response to PRM Issue 2.

III. Public Comments on the Petition.

The NRC received three public comment submissions on the PRM, one each from the following: the Nuclear Energy Institute (NEI), Exelon Generation Company, and the petitioner. In addition to those submissions, the NRC received a late-filed comment submission from the petitioner in response to the NEI comment submission. The late-filed comment submission, submitted by the PRM-50-105 petitioner, contains some reiteration of information and assertions in PRM-50-105. The NRC is not addressing those portions of the late-filed comment response. However, the late-filed comment submission also discussed matters related to the use of in-core thermocouples in gamma thermometers, the use of in-core thermocouples in the Economic Simplified Boiling Water Reactor (ESBWR) design, and the radiation dose to workers due to in-core thermocouples; these issues were not raised in the original PRM. Therefore, the NRC is addressing these three new matters in this comment response section.

The comments are grouped into four comment categories: General Discussion of PRM-50-105, Comments on In-Core Thermocouples, Comments Related to Westinghouse AP1000, and Comments on Experimental Data. A comment identifier (e.g., NEI-1) follows each comment summary. The comments and the associated NRC responses follow.

General Discussion of PRM-50-105

Comment: The NRC should not amend its regulations to require all holders of operating licenses to operate nuclear power plants with in-core thermocouples at different elevations and radial positions throughout the reactor core. (NEI-1)

NRC Response: The NRC agrees with this comment. The NRC is denying PRM-50-105 for the reasons set forth in this document.

Comments on In-core Thermocouples:

Comment: Use of in-core thermocouples would result in higher doses to workers both to implement plant modifications and to maintain the proposed system with minimum if any benefit to plant safety. (NEI-2)

NRC Response: The NRC agrees with the comment, but notes that the comment did not provide any basis for this assertion.

Comment: In response to another commenter's statement that in-core thermocouples would result in a higher radiation dose to workers both to implement plant modifications and to maintain the proposed system with minimum, if any, benefit to plant safety, one commenter provided the following quote from General Electric Hitachi (GEH) Nuclear Energy: "A [gamma thermometer] system has no moving parts, no under vessel tubing, virtually no radiation dose to maintenance since it is a fixed in-core probe, and is expected to be very reliable."[6] The commenter asserts that in-core thermocouples could be placed inside instrument tubes, distributed through the reactor core, like gamma thermometers are, and thus cause virtually no radiation dose to workers during maintenance. (Leyse2-5)

[6] GE Hitachi Nuclear Energy, "Licensing Topical Report: Gamma Thermometer System for [Local Power Range Monitor] LPRM Calibration and Power Shape Monitoring," NEDO-33197-A, p. 1 (available at ADAMS Accession No. ML102810320).

NRC Response: The NRC disagrees with the comment that in-core thermocouples would cause virtually no radiation dose to workers during maintenance. The NRC notes that the GEH report, referenced by the PRM as support for the comment, applies only to a comparison of the current BWR moveable and retractable probe (the TIP system) with the ESBWR fixed incore gamma thermometers. It does not apply to the installation of in-core thermocouples in currently operating reactors. The NRC agrees that the use of fixed versus bottom entry retractable sensors may reduce exposure for routine maintenance. The NRC continues to believe that in-core thermocouples would result in a higher radiation dose to workers while implementing the necessary plant modifications for installation and to maintain the proposed system, particularly when replacement of sensor strings due to long-term radiation exposure is required. Also, except for existing tubing for bottom-entry removable sensors, any existing instrument tubes are already occupied. It is likely that new instrument tubes would need to be installed. Tubes installed through the vessel head would also require provisions for mechanical and electrical connections. These installation efforts, whether the new tubing enters the core through the vessel head or bottom, are likely to require significant worker exposure, and may also raise concerns related to pressure boundary integrity.

Comment: In some designs, in-core thermocouples could be more susceptible to failures and misdiagnosis than CETs because of proximity to thermal and radiation sources. It is not feasible to attach thermocouples directly to the fuel cladding. Thermocouples would need to be located in existing instrument tubes (e.g., BWR Local Power Range Monitor tubes) and would not be in direct contact with the reactor coolant. Therefore, thermocouples would provide only indirect readings of fuel temperature and would be subject to heat transfer delays/response times. The time response and accuracy of the reading as it relates to the reactor coolant would be highly questionable. The presence of the fuel channel on a BWR fuel assembly would further inhibit and interfere with the readings of a thermocouple in an instrument tube. (NEI-3) (Exelon-2)

NRC Response: The NRC acknowledges that in-core thermocouples could be more susceptible to failure and misdiagnosis in some designs. However, as stated throughout this document, because CETs perform their desired functions and because precise knowledge of in-core temperatures would not change operator actions, further consideration of the potential limitations of in-core thermocouples is not necessary.

Comment: In response to another commenter's assertion that in-core thermocouples may be more susceptible to failures and misdiagnosis than CETs, one commenter stated that in-core thermocouples have been tested and used in nuclear reactors for decades as the primary component of in-core gamma thermometers (devices that measure gamma flux in nuclear reactors). Radcal gamma thermometers were installed in PWRs in the 1980s. Radcal thermometers are also installed in BWRs. General Electric Hitachi Nuclear Energy has plans to use in-core thermocouples in gamma thermometers in the ESBWR design. (Leyse2-1) (Leyse2-2) (Leyse2-4)

NRC Response: The NRC continues to believe that CETs are acceptable for use in current applications. Where current nuclear power plants have fixed in-core gamma thermometers, they are for power shape monitoring and calibration, not for actual temperature measurements. Further, the gamma thermometer GEH plans to install in the ESBWR is a device for measuring the gamma flux for the purpose of calibration of the local power range monitors and power shape monitoring; the gamma thermometers are *not* for the purpose of measuring axial and radial core temperature. The GEH gamma thermometers utilize a local differential temperature directly within the sensor at the specific sensor location to infer the gamma flux inside the reactor core rather than the actual temperature measurements at that location. Actual temperature measurements are not available outside the reactor core. For these reasons, the information about the use of gamma thermometers at nuclear power reactors and in the ESBWR design certification do not affect the NRC's position that CETs are acceptable for use in current applications to perform their specified function.

Comment: An Idaho National Laboratory (INL) report stated that INL "developed and evaluated the performance of a high temperature resistant thermocouple that contains doped molybdenum and a niobium alloy. Data from high temperature (up to 1500 °C), long duration (up to 4000 hours) tests and on-going irradiations at INL's Advanced Test Reactor demonstrate the superiority of these sensors to commercially-available thermocouples. However, several options have been identified that could further enhance their reliability, reduce their production costs, and allow their use in a wider range of operating conditions."[7] (Leyse2-3)

NRC Response: The information in the comment is not relevant to the PRM, and therefore does not change the NRC's position that CETs are acceptable for use in performing their specified function, thereby obviating the need to install in-core thermocouples. The NRC also notes that the pre-publication INL report dated 2009 referenced by the commenter described a research product that is not yet ready for commercial use by the nuclear industry. The NRC does not believe that the statements in the report that are referenced in the comment are relevant to the acceptability of CETs in current applications.

Comment: The transition from EOPs to SAMGs based on existing plant parameters is adequate. Pressurized Water Reactors already use CETs to make the transition to SAMGs.

---

[7] Joshua Daw, et al., Idaho National Laboratory, "High Temperature Irradiation-Resistant Thermocouple Performance Improvements," INL/CON-09-15267, Sixth American Nuclear Society International Topical Meeting on Nuclear Plant Instrumentation, Control, and Human-Machine Interface Technologies, April 2009, p. 1 (available at http://www.inl.gov/technicalpublications/documents/4235634.pdf).

The potential delay in the response of indirectly reading in-core thermocouples could actually be longer than the response of other plant parameters, including CETs, in identifying potential severe accident conditions. (Exelon-3)

NRC Response: The NRC agrees that the current transition from EOPs to SAMGs is adequate. The NRC notes that SAMGs are developed based on the recognition that CETs could differ from actual core temperatures. This concept is described in Section II, “NRC Technical Evaluation,” of this document.

Comment: During steady-state operations for both PWRs and BWRs, the fuel cladding (surface) temperature is a function of coolant Temperature – Enthalpy (T-H) properties. The coolant steady-state properties (i.e., temperature) do not vary significantly axially or radially during steady-state operation and therefore, in-core thermocouples would not provide useful information. There are more accurate means of measuring core conditions than in-core thermocouples already in place. Adding in-core thermocouples would not improve the ability or accuracy of measuring core conditions. (Exelon-1)

NRC Response: The NRC agrees with the comment. The PWR in-core conditions, for example, are measured using hot and cold leg temperatures, reactor coolant pressure, and neutron flux. These parameters are then used as inputs to the reactor protection system to ensure that the reactor shuts down if core operating conditions deviate significantly from the expected normal operating conditions. The BWRs are equipped with similar equipment intended for monitoring normal, steady-state operation. The addition of in-core thermocouples, either to measure fuel surface or reactor coolant temperatures, would add little value to the information already available for monitoring normal operation.

Comment: The petitioner asserted that, in the event of a severe accident, in-core thermocouples would provide nuclear power plant operators with "crucial information to help operators manage the accident." However, the petitioner provided no basis that actions taken by operators would be more effective than actions based on existing CETs. Operators are trained to recognize off-normal operating conditions that have potential for resulting in core damage and to maneuver the plant to a more conservative operating envelope (i.e., provide coolant to the reactor core). In a severe accident, operator strategies control parameters across large regions of the core or across the entire core. The additional information regarding local fuel temperature provided by in-core thermocouples would not be crucial relative to restoring coolant, nor would it change the steps and actions available to operators to maintain or restore adequate core cooling conditions. There is no evidence to show that temperatures sensed at a single location could be used more effectively than actions based on CET temperatures. (Exelon-4) (NEI-4) (NEI-6)

NRC Response: The NRC agrees with the comment. Precise measurement of local fuel temperatures at distinct locations throughout the core would not provide essential data for informing severe accident management decisions, and the petitioner cited no actions that would be driven by the additional information obtained from in-core thermocouples. In the event of an extended loss of core cooling that leads to core damage, the actions taken by the operators will be focused on restoring core cooling, with or without the knowledge of precise fuel temperatures in the core.

Comments Related to Westinghouse AP1000:

Comment: One commenter provided several comments on the emergency response guidelines for Westinghouse's AP1000 design:

- Westinghouse maintains that core exit gas temperature would reach 1200 °F in Time Frame 1, but the LOFT LP-FP-2 experiments show that core exit temperatures were measured at around 800 °F when in-core thermocouples measured fuel cladding temperatures exceeded 3300 °F. Thus, after the onset of the rapid zirconium-steam reaction, core exit temperatures were measured at around 800 °F. (Leyse-4)

- There are problems with Westinghouse's emergency response guidelines for the AP1000. Plant operators are instructed to actuate the AP1000 containment hydrogen igniters after the CET measurements exceeded 1200 °F, which would most likely be some time after a meltdown had commenced. (Leyse-6)

- There are problems with Westinghouse's plan to have plant operators rely on CET measurements in the event of a severe accident, because plant operators might reflood an overheated core without realizing that the core was in fact overheated. Consider a scenario where there were similar temperature differences between in-core and core exit temperatures as were observed in LOFT LP-FP-2. If plant operators were to reflood the core when core exit temperatures were well below 1200 °F, the core could already be overheated (i.e., fuel-cladding temperatures could be over 3300 °F), nearing the temperature where zirconium melts. In such a case there would also be some liquefaction of core components because of eutectic reactions (i.e., the eutectic reaction between zirconium and stainless steel) taking place at temperatures as low as 2200 °F. Unintentionally reflooding an overheated core could be very dangerous. In a severe accident, during the reflooding of an overheated reactor core up to 300 kilograms of hydrogen could be generated in one minute. (Leyse-7)

- It is evident that with Westinghouse's plan to have plant operators rely on CET measurements in the event of a severe accident, operators could unintentionally reflood an overheated core, which would rapidly generate additional hydrogen, at a rate as high as 5.0

kilograms per second, which could, in turn, compromise the containment if the hydrogen were to detonate. (Leyse-8)

- For severe accidents, Westinghouse's plan for AP1000 plant operators to rely on core exit temperature measurements to monitor the condition of the core and to wait for a core exit temperature measurement of 1200 °F to signal when to actuate the hydrogen igniters and implement other procedures would be neither productive nor safe. (Leyse-10)

NRC Response: The NRC disagrees with the comments that the Westinghouse emergency response guidelines for the AP1000 design are inadequate, based upon CET limitations. As discussed throughout this document, the CET limitations noted in both this comment and the PRM are acknowledged by the NRC and have been documented in industry reports. The CETs, even with their known limitations, are sufficient to provide the necessary information to nuclear power plant operators. More precise knowledge of in-core temperatures would not change the operational decisions necessary in the event of a severe accident. Therefore, the NRC does not believe that the comment provided information supporting the PRM's request that nuclear power plant licensees be required by rule to install in-core thermocouples.

To the extent that the comments raise issues with respect to the adequacy of the AP1000 design and hydrogen control, the NRC regards this portion of the comment to be outside the scope of the issues raised in this PRM. The NRC notes, however, that these AP1000 issues were raised in a 10 CFR 2.206 petition on Vogtle, Units 3 and 4 (ADAMS Accession No. ML12061A218), and resolved as part of the NRC's action on the petition. The NRC's resolution of the 10 CFR 2.206 petition is available at ADAMS under Accession No. ML13105A308.

Comments on Experimental Data:

Comment: The commenter cited the OECD Nuclear Energy Agency report, which states: "During the rapid oxidation phase [core exit temperatures] appeared essentially to be disconnected from core temperatures." (Leyse-5)

NRC Response: The following sentence appears in the same section of the OECD report referenced by the commenter: "For core runaway conditions with rapid fuel oxidation, LOFT results indicated that the CETs essentially were disconnected from the core temperatures. This is perhaps a lesser problem since such conditions cannot be well addressed by accident management measures." Currently, CET indications are used to help determine core uncovery and initiate appropriate actions during that phase of an accident. In following phases, core temperatures do not provide information that is used to initiate actions to mitigate an accident.

Comment: Two of the main conclusions from data from experiments simulating design basis accidents conducted at four different facilities are that core exit temperature measurements display in all cases a significant delay (up to several hundred seconds) and that core exit temperature measurements are always significantly lower (up to several hundred degrees Celsius) than the actual maximum cladding temperature. (Leyse-9)

NRC Response: The NRC agrees with this comment. The NRC was directly involved in most of the experimentation referenced by the petitioner, and the NRC and other nuclear industry stakeholders have been aware for several years of the CET limitations concluded from the experiments and verified by independent analyses. Evidence of this can be seen in WCAP-14696-A, Revision 1 (November 1999; ADAMS Accession No. ML993490267), which states that "Analyses performed for the WOG [Westinghouse Owners Group] ERGs [Emergency Response Guidelines] for indication of inadequate core cooling concluded that the temperature indicated by the core exit thermocouples, especially during transient heat up conditions, is always several hundred degrees lower than the fuel rod cladding temperatures." The NRC notes that SAMGs are developed based on the recognition that CETs could differ from actual core temperatures. This concept is described in Section II, "NRC Technical Evaluation," of this document.

Miscellaneous Comments:

Comment: An April 2012 Advisory Committee on Reactor Safeguards (ACRS) report states that the NRC "has recognized the need for enhanced reactors…instrumentation and is in the process of adding this to the implementation of the NTTF [Near-Term Task Force] recommendations." And the NTTF report "recommends strengthening and integrating onsite emergency response capabilities such as EOPs and SAMGs." The April 2012 ACRS report states that "such integration could focus on the need to clarify the transition points" that would occur in a NPP accident. In-core thermocouples would fulfill the need for enhanced reactor instrumentation. In-core thermocouples would provide NPP operators with crucial information to help them track the progression of core damage and manage an accident; for example, indicating the correct time to transition from EOPs to implementing SAMGs. (Leyse-1)

NRC Response: The NRC disagrees with this conclusion. As stated previously in this document, at no point, neither during diagnosis nor follow-on actions to restore cooling, is there an operational necessity for an exact measurement of core temperatures at various locations throughout the core. However, as noted in Enclosure 3 to SECY-12-0095, "Tier 3 Program Plans and 6-month Status Update in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Subsequent Tsunami," dated July 13, 2012 (ADAMS Accession No. ML12208A210), the NRC indicated that it added the ACRS recommendation that "Selected reactor and containment instrumentation should be enhanced to withstand beyond-design-basis accident conditions” to the Tier 3 activities implementing a set of the NRC’s Near-Term Task Force (NTTF) recommendations (*Recommendations for Enhancing Reactor Safety in the 21st Century*, dated July 12, 2011, ADAMS Accession No. ML112510271). The scope of the Tier 3 long-term evaluation is much broader than, and does not focus on, the use of thermocouples. Rather, the Tier 3 evaluation will focus on the entire suite of instrumentation available to operators during a beyond-design-basis accident.

Comment: BWRs need to operate with in-core thermocouples and noted the following:

- CETs are not installed in BWRs. In the event of a severe accident, BWRs are supposed to detect inadequate core cooling and core uncovery by measuring the water level in the reactor core. However, “BWR high drywell temperature and low pressure accidents ([for example,] LOCAs) can cause the water level to read erroneously high…and BWR water level readings are unreliable after core damage.” (Leyse-2a)
- By the time BWR operators confirm an accelerated core melt (by measuring increased reactor and containment pressure rates and/or wetwell water temperature rises), the reactor core would already be overheated and reflooding an overheated core could generate hydrogen, at rates as high as 5.0 kg per second. (Leyse-2b)
- In the event of a BWR severe accident, in-core thermocouple measurements would be more accurate and immediate for detecting inadequate core cooling and core uncovery than readings of the reactor water level, reactor pressure, containment pressure, or wetwell water temperature. (Leyse-3)

NRC Response: The NRC considers this comment to be outside the scope of the matters raised in the PRM. As discussed at the beginning of the NRC’s technical evaluation of this PRM, and in “PRM Issue 2: Nuclear Power Plant Operators’ Use of In-Core Thermocouples,” the NRC is evaluating the PRM as it pertains to PWRs only for the reasons indicated in those sections. Furthermore, the section addressing PRM Issue 2 describes some challenges with the use of in-core thermocouples, both surface-mounted thermocouples and thermocouples in bulk coolant areas. Those challenges would exist in BWR applications, as well.

Comment: The proposed additional instrumentation is relevant only to postulated core conditions where CETs indicate some small amount of sub-cooling while in-core thermocouples indicate locally higher temperatures with less sub-cooling. Where CET sub-cooling is minimal, operators are trained to take actions to increase this margin. Existing procedures and a predetermined CET value concurrently provide adequate indication for plant operators to transition from EOPs to implementing SAMGs. (NEI-5)

NRC Response: The NRC agrees with the comment. As stated in response to comments Exelon-4/NEI-4/NEI-6 and Leyse-5, operator actions are not focused on localized core conditions. Rather, actions are based on bulk CET readings. These readings are established in consideration of expected differences between local conditions and the associated CET conditions.

IV. Ongoing NRC Activities Related to Reactor and Containment Instrumentation.

As noted in the “Miscellaneous Comments” subsection of Section III of this document, the NRC has added the ACRS recommendation that “Selected reactor and containment instrumentation should be enhanced to withstand beyond-design-basis accident conditions” to the Tier 3 activities implementing a set of the NRC’s NTTF recommendations. The scope of the Tier 3 long-term evaluation will focus on the entire suite of instrumentation available to operators during a beyond-design-basis accident. These activities will support decisions on whether there is a need for subsequent regulatory action, including rulemaking, in that area. If the NRC decides that rulemaking is necessary in the area of reactor instrumentation, the public will have an opportunity to provide comments as part of publication of a proposed rule in the *Federal Register*.

V. Availability of Documents.

The following table provides information on how to access the documents referenced in this document. For more information on accessing ADAMS, see the ADDRESSES section of this document.

Date	Document	ADAMS Accession Number/*Federal Register* Citation/URL
February 28, 2012	Incoming Petition (PRM-50-105) from Mr. Mark Leyse	ML12065A215
May 23, 2012	Mr. Mark Leyse; Notice of Receipt of Petition for Rulemaking	77 FR 30435
November 26, 2010	Organisation de Cooperation et de Developpement Economiques; "Core Exit Temperature (CET) Effectiveness in Accident Management of Nuclear Power Reactor (NEA/CSNI/R(2010)9)"	http://www.oecd-nea.org/nsd/docs/2010/csni-r2010-9.pdf
1963	Dougall, R. S. and W. M. Rohsenow, "Film Boiling on the Inside of Vertical Tubes with Upward Flow of the Fluid at Low Qualities"	http://hdl.handle.net/1721.1/62142
January 1, 1974	Adams, J. P. and G. E. McCreery, "Limitations of Detecting Inadequate Core Cooling"	http://www.osti.gov/energycitations/product.biblio.jsp?osti_id=6797561

November 1999	WCAP-14696-A, Revision 1, "Westinghouse Owners Group Core Damage Assessment Guidance"	ML993490267
November 1980	NUREG-0737, "Clarification of TMI Action Plan Requirements"	ML051400209
July 13, 2012	Enclosure 3 to SECY-12-0095, "Tier 3 Program Plans and 6-month Status Update in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Subsequent Tsunami"	ML12208A210
October 2010	Licensing Topical Report, "Gamma Thermometer System for LPRM Calibration and Power Shape Monitoring"	ML102810320
April 2009	Idaho National Laboratory, "High Temperature Irradiation-Resistant Thermocouple Performance Improvements"	http://www.inl.gov/technicalpublications/documents/4235634.pdf
February 28, 2012	2.206 Petition on Vogtle, Units 3 and 4	ML12061A218
April 30, 2013	Closure Letter to Mr. Mark Leyse re. 2.206 Petition on Vogtle, Units 3 and 4	ML13105A308
July 12, 2011	Recommendations for Enhancing Reactor Safety in the 21st Century	ML112510271
August 2, 2012	Comment Submission (1) from Nuclear Energy Institute	ML12216A082
August 6, 2012	Comment Submission (2) from Mr. Mark Leyse	ML12219A362
August 7, 2012	Comment Submission (3) from Exelon Generation	ML12230A296

August 22, 2012	Comment Submission (4) from Mr. Mark Leyse	ML12237A263

VI. Determination of the Petition.

During normal operation in a PWR, RCS hot leg and cold leg temperatures are the primary indications of core condition. Measurements of RCS hot and cold leg temperatures from safety-related instrumentation provide the necessary input to a plant's reactor protection system. There are no reactor protection or plant control functions that use inputs from the CETs. Additionally, the CETs are not the only source of information relied on to initiate reactor operator responses to accident conditions.

The NRC has determined that there is no operational necessity for an exact measurement of core temperatures at various locations throughout the core. The petitioner provided no justification why the precise knowledge of core temperature would enhance safety or change operator actions during normal or accident conditions. Furthermore, there are no reactor protection or plant control functions that use inputs from the CETs.

Contrary to the petition's assertion that an OECD report supports a determination that CETs have limitations, the NRC notes that the same OECD report stated that "despite the delay and the difference in the measured temperatures, the time evolution of the CET signal readings in the center section seem to reflect the change of the cooling conditions in the core and thus the tendency of the maximum cladding temperatures quite well." The NRC acknowledges the limitations of CETs but concludes that CETs are sufficiently accurate to support appropriate operator action in a timely fashion during an accident. The NRC's conclusion is consistent with the conclusions of various industry organizations that the use of CETs is appropriate and safe.

For these reasons, the NRC declines to undertake rulemaking to require installation and use of in-core thermocouples. Accordingly, the NRC is denying PRM-50-105 in accordance with 10 CFR 2.803. The NRC's decision to deny the PRM included consideration of public comments received on the PRM.

Dated at Rockville, Maryland, this 6th day of September, 2013.

For the Nuclear Regulatory Commission.

/RA/

Richard J. Laufer,
Acting Secretary of the Commission.

[FR Doc. 2013-22234 Filed 09/11/2013 at 8:45 am; Publication Date: 09/12/2013]